Neutronic study of slightly modified water reactors and application to transition scenarios

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In this paper we have studied slightly modified water reactors and their applications to transition scenarios. The PWR and CANDU reactors have been considered. New fuels based on Thorium have been tested : Thorium/Plutonium and Thorium/Uranium-233, with different fissile isotope contents. Changes in the geometry of the assemblies were also explored to modify the moderation ratio, and consequently the neutron flux spectrum. A core equivalent assembly methodology was introduced as an exploratory approach and to reduce the computation time. Several basic safety analyses were also performed.
We have finally developed a new scenario code, named OSCAR (Optimized Scenario Code for Advanced Reactors), to study the efficiency of these modified reactors in transition to GenIV reactors or in symbiotic fleet.

I INTRODUCTION

Nuclear energy allows the production of electricity almost free of CO_2 emissions and thus is a desirable technology for combatting climate change. Moreover, fossil fuel natural resources are expected to decrease significantly within a few decades. Thus, the fraction of global energy provided by nuclear power is expected to increase significantly before the end of this century. If such a large expansion takes place, the access to cheap uranium natural resources may be severely limited in the future. This is one of the problems the new generation (GenIV) of nuclear reactors, currently under development, has to face. These reactors are designed to be safe and use resources more efficiently. The use of Thorium is also studied for several designs.

Several scenarios Ref. [1, 2, 3] have been proposed with the eventual goal of reaching slightly converting or full breeding cycles with technologies such as Fast Breeder Reactors (FBRs) or Molten Salt Reactors (MSRs).

In these studies, the transition between current nuclear plants (mainly LWRs and some HWRs) and the fourth generation reactors is assumed to be direct, i.e. actual designs using conventional fuels are replaced by GenIV reactors as fast as the necessary fissile fuel stocks can allow it. Obtaining a self-sustainable reactor fleet is a paramount goal. However, the choices made to achieve the transition may have a large impact on the total amount of nuclear waste produced, or simply on the feasability of the transition itself. In this perspective, slightly modified water reactors would be an excellent option to help optimize such a transition, without too high extra costs.

In this study, we have first prospected a large panel of slight modifications on water reactors at elementary cell and assembly levels. Afterwards, the integration of these reactors in transition scenario is investigated. The practical application of the French nuclear plant fleet is proposed.

II CODES AND METHODOLOGY

II.A Codes

To deal with exotic core designs (with different moderation ratios for instance) and exotic fuels (mainly Th/U-233 and Th/Pu), confrontation of two different codes should be necessary in order to gain confidence in our results despite a crual lack of experimental data. We use :

- MURE Ref. [4] (MCNP Utility for Reactor Evolution), a French (CNRS) C++ tool for nuclear reactor reference calculations interfacing the particle transport code MCNP Ref. [5].

- DRAGON Ref. [6] (transport method), a deterministic Canadian cell code for lattice cell calculations and reactivity device calculations and DONJON Ref. [7] (diffusion method), a finite core code we used for neutron leakage and dwell time evaluations. These codes allow us to quickly define reactor design and estimate fuel performance.

Nuclear data used are ENDF/B-VI (release 8) for MURE and ENDFB6 (the 172-group WIMS-D library from the IAEA WLUP web site) for DRAGON.

To avoid any too long transport calculation of the core, the calculation scheme uses a core-equivalent assembly which stands for the core average neutronic properties.

II.B Reactor designs and calculation principles

II.B.1 *PWR*

A N4 PWR core of 205 assemblies is simulated, with a 17*17 fuel assembly composed of 264 fuel pins and 25 guide tubes replaced by water holes. Temperatures are 575K for cladding tubes and moderator H_2O and 1000K for the fuel itself. Boron influence has been checked to be negligible compared to other factors like nuclear data uncertainties (relative differences on main inventories up to 4%). For definitive results, no burnable material is thus included in our assembly descriptions. The fuel loading enrichment is homogeneous in the assembly.

The computational scheme for our PWR core-equivalent assembly consists of four stages (Fig. 1) :

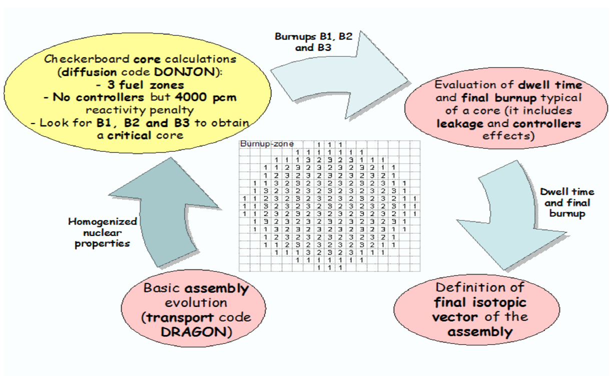

FIG. 1 – Computational scheme for PWR studies

1. Basic assembly evolution and calculation of the average nuclear properties (with DRAGON, cross-checked with MURE).
2. Core instantaneous time calculations with DONJON from homogenized 2-group assembly cross-sections provided by DRAGON : a checkerboard core model is used with three fuel compositions at different burnups (B1, B2 and B3 see on Fig. 2). Reflector data are computed with a very simplified core transport calculation. A 4000 pcm Ref. [8] reactivity penalty stands for controllers.
3. Evaluation of dwell time, final burnup and neutron leakage from DONJON results.
4. Application of dwell time and final burnup on the lattice cell results to define the final isotopic vector and the global conversion ratio from both DRAGON and MURE results.

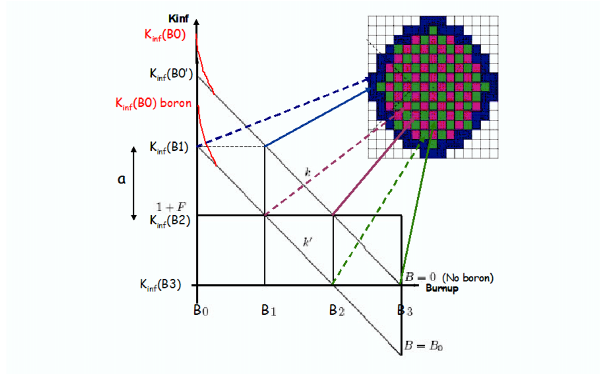

FIG. 2 – Hypotheses and notations on our PWR model k_{inf} evolution regarding boron

Fig. 2 is an illustrative outline of the typical k_{inf} evolution that we assumed in our model. This scheme implies two main hypotheses namely, k_{inf} linearity and negligible boron impact on the fuel isotopic vector. At the beginning of the cycle, a fast k_{inf} drop can be seen due to the Xenon saturation. A calculation with boron is performed to verify that a core calculation at the beginning of the cycle is achievable and to control that corresponding leakage is consistent with the value obtained without boron. The way to compute initial k_{inf} with boron and thus boron amount is :

$$k_{eff}(EOC) = 1 + F = \frac{k_{inf}(B1)+k_{inf}(B2)+k_{inf}(B3)}{3}$$
$$\text{and } a = \frac{k_{inf}(B1)-k_{inf}(B3)}{2}$$

$$k_{inf}(B0') = a + k_{inf}(B1) \text{ and}$$
$$k_{inf}(B0)_{boron} = 1 + F + a + k_{inf}(B0) - k_{inf}(B0')$$

$$k_{inf}(B0)_{boron} = 1 + F + k_{inf}(B0) - k_{inf}(B1)$$

For each case studied, assembly power distribution and form factor are checked against intra-assembly power peaking.

Fuels selected for analysis are UOX with 3.5 % U-235, Th/Pu a mixed Plutonium-Thorium dioxide (Plutonium from a N4 PWR, described by Table 1) and Th/U-233 (Thorium-Uranium 233).

TABLE 1 – Used Plutonium isotopic vector

Isotope	Pu^{238}	Pu^{239}	Pu^{240}	Pu^{241}	Pu^{242}
% (at)	3.1	52.5	24.5	12.2	7.7

Final burnups met by means of different fuel enrichments range between 12000 and 66000 MWd/t heavy metal (HM) (between one year and four years compared to around 3 years for a UOX N4 PWR).

Moderation ratio (MR) defined as the ratio of moderator volume over fuel volume can be modified through the assembly lattice pitch . The fuel rod diameter and the cladding thickness are kept identical to the N4 standard. Contrary to water holes, fuel pins have a variable number. The heat removed per pin by water has to remain close to the heat usually removed (to remain in known thermal-hydraulic parameters) so power per pin is chosen identical to those of a UOX assembly. Hence variable MR makes core power change, while core size remains constant.

II.B.2 *CANDU*

The CANDU-6 reactor is a heavy-water-moderated and cooled reactor using natural uranium as a fuel. The CANDU design is modular, with 380 fuel channels on a square lattice. Each fuel channel contains 12 fuel bundles composed by 37 pins of 50 cm long (Fig. 3). Temperatures are 350K for steel vessel, moderator D2O and calandria tubes, 575K for coolant, pressure tubes and fuel clads, and 900K for fuel. To keep the reactor critical, bundles are regularly on-power refueled.

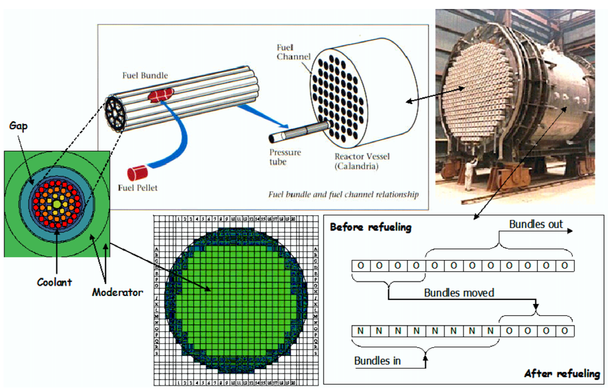

FIG. 3 – CANDU-6 core structures and typical refueling schemes

The primary goal is to obtain a dwell time which matches with CANDU technology limits namely refueling capacities (low burnups lead to a high refueling frequency) and pressure (and calandria) tube behaviour (high burnups can damage cladding under irradiation). Thus dwell time and final burnup selection determine fuel enrichment. Selected burnups are included between 2500 and 22000 MWd/t HM (natural Uranium fuel burnup being around 7600 MWd/t HM).

The methodology is similar to PWR one : bundle evolution with the transport code then core calculation and leakage/dwell time evaluation. Due to the online refueling, divergences occur on core calculations : the time-averaged approach (calculations are performed at refueling equilibrium using cross-sections averaged on the fuel dwell time) is applied to compute burnup values. Two fuel zones are described, controller devices such as zone controllers and adjusters which are positioned interstitially between fuel channels are simulated, and reflector zone properties are taken identical to bundle moderator ones. Such a simulation of the realistic fuel management allows to reproduce actual reactor core configurations and to obtain a flat enough power core distribution.

Bundle MR is changed through the lattice pitch with a constant fuel volume. Core is modeled to keep the same number of channels and bundles and thus core size depends on the MR. Then controller devices volume also varies with the MR. In order to maintain controller devices efficiency, a corrective factor is introduced to adjust their macroscopic cross-sections. Besides, thermal-hydraulic parameters are assumed not to change.

III PHYSICAL AND PARAMETRICAL STUDY

The objectives of this parametrical study are to define a feasibility area with regard to the following neutronic aspects : MR, fuel loading, dwell time and feedback safety coefficients. Design goals depend on criteria selected in nuclear scenarios : environmental concerns and waste management, increasing of energy self-sustainability, fuel ressource management strategies, interest in actinide burning, economics and other national policies. So a lot of various priorities could be applied. In this non-exhaustive analysis, high conversion ratio assemblies and U-233 breeders have been particularly investigated.

III.A Neutronic Analysis and Conversion Ratio

To rank competitiveness of once-through reactor/fuel systems, the Conversion Ratio (CR), defined as a function of the fissile (fissiles considered are U-235, Pu-239, Pu-241 and U-233+Pa-233) and heavy nuclei mass inventory variations between Beginning Of Cycle (BOC) and End Of Cycle (EOC), is used :

$$CR = 1 - \frac{\Delta FissileNucleiMass}{\Delta HeavyNucleiMass}$$

Heavy nuclei mass variation being imposed by power, fissile nuclei mass variation has to be optimized to achieve maximum fuel efficiency.

First a modification of fissile requirements is assessed. The higher parasitic capture rates in the PWR lattice (compared to the excellent neutron economy of CANDU) need to be compensated by extra fissile material.

Figures 4 and 5 (MURE results) show that a lower fuel enrichment allows to enhance the CR. Indeed minor actinide production and fission product build-up increase with burnup and lead to an accumulation of non-fissile neutron absorbers. Furthermore, as explained earlier, dwell time cannot be too short. For each reactor, a compromise must be found between dwell time and fuel fissile enrichment.

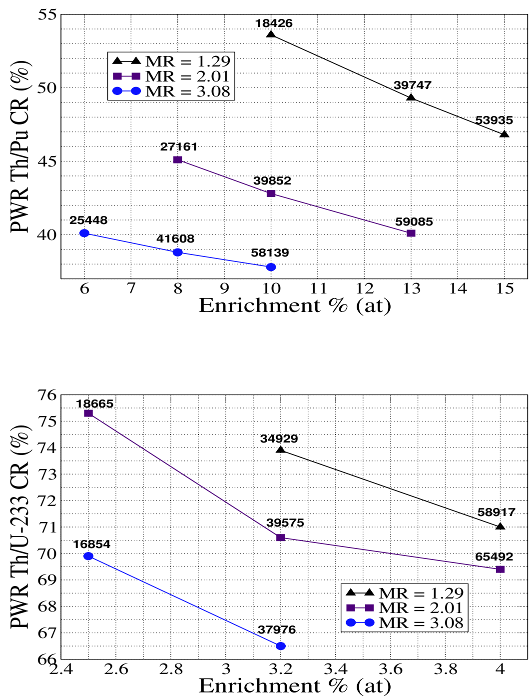

Fig. 4 – Conversion Ratio vs. fuel enrichment for PWR fuels with various MRs (numbers above points are final burnups in MWd/t HM)

Next, in order to reach higher CR, changes of MR are examined. Apparently PWR and CANDU core designs offer a great range of possibilities to be modified, but for PWR previous studies Ref. [9] established that MR can hardly be lower than 0.8 because of heat extraction issues. Some thermal-hydraulic and mechanic parameters considerations have to be taken into account for CANDU as well. Thus just one case of over-moderation and under-moderation for each reactor are presented here. It underlines a trend but it does not consist of an optimization.

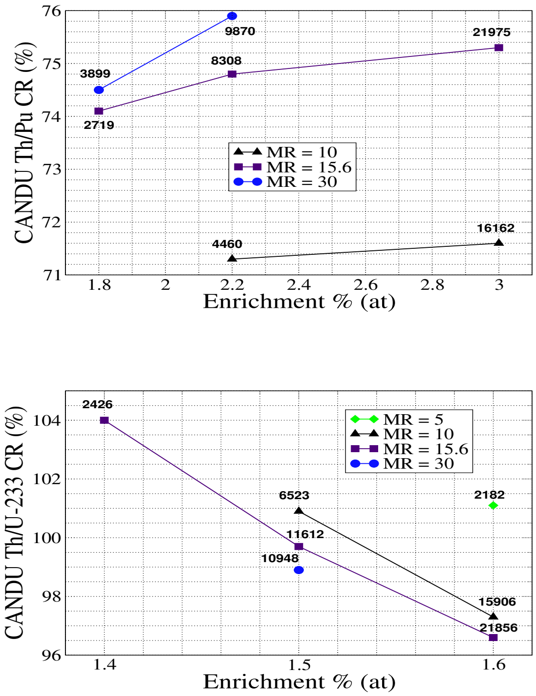

Fig. 5 – Conversion Ratio vs. fuel enrichment for CANDU fuels with various MRs (numbers above points are final burnups in MWd/t HM)

As illustrated in these graphs, in PWR, a harder neutron spectrum is required to improve the CR. As a matter of fact, lower MRs reduce the resonance escape probability, increasing the Th-232 fertile capture rate. However, even if fuel enrichment is reduced, the upper limit of the CR is 0.75 in the assembly (if we want that dwell time remains above one year). It is well-known Ref. [10] that higher CR could be achieved with fertile blankets in a heterogeneous core.

As for CANDU, neutron spectrum variations are minor given that CANDU neutron spectrum is more thermal. CR evolution with MR depends on fuel type. The Th/Pu CR is higher with over-moderation and inversely for the Th/U-233 fuel. If further improvements are made to the CANDU neutron economy (such as removal of adjuster rods and use of enriched zirconium for structural materials), its CR could be slightly increased and so the self-sufficient Thorium cycle would be probably feasible, even with fuel multi-recycling.

We use another factor to compare Th/Pu fuels :

$$CF = \frac{(m_{U233}+m_{Pa233})^{out}}{m_{Pu}^{in}} * 100(\%)$$

This Conversion Factor (CF) estimates the U-233 production potential of assemblies. Results are shown on Tables 2 and 3. Higher is this conversion factor, higher is the U-233 production.

TABLE 2 – MURE results for Th/Pu PWRs

Pu (at %) / MR	$BU^{(+)}$	CR	$CF(*)$
10% / 3.08	58139	37.8	13.2 (14.1)
10% / 2.01	39851	42.8	12.4 (28.7)
10% / 1.29	18426	53.6	8.5 (42.8)
13% / 2.01	59084	40.1	11.8 (26.0)
13% / 1.29	39746	49.3	11.2 (37.2)
15% / 1.29	53935	46.8	11.6 (34.7)
8% / 3.08	41608	38.8	14.1 (17.7)
8% / 2.01	27161	45.1	12.4 (31.9)
6% / 3.08	25447	40.1	14.3 (23.8)

$^{(+)}$ (MWd/t)

$(*)$ numbers in brackets represents the Pu-239 percent proportion in the Pu isotopic vector at EOC

TABLE 3 – MURE results for Th/Pu CANDUs

Pu (at %) / MR	$BU^{(+)}$	CR	$CF(*)$
1.8% / 30	3899	74.5	17.7 (39.8)
1.8% / 15.6	2719	74.1	12.7 (43.6)
2.2% / 30	9870	75.9	29.3 (26.7)
2.2% / 15.6	8308	74.8	25.7 (30.5)
2.2% / 10	4460	71.3	15.2 (40.6)
3.0% / 15.6	21975	75.3	33.2 (14.7)
3.0% / 10	16162	71.6	28.1 (22.9)

These tables indicate that a PWR core exhibits behaviour similar to a CANDU core with respect to the effect of MR modifications on the CF. Indeed over-moderation is required to improve this conversion factor.

So a highly enriched and/or over-moderated CANDU core seems to be an efficient means of producing a stockpile of valuable U-233.

III.B Reactor and Safety Parameters

Core concepts aforementioned appear as alternative reactors for transition scenarios from GenIII to GenIV. But a Th/Pu fueled reactor will not behave the same as a UOX one or a Unat one. Safety parameters like Doppler coefficient, Coolant Void Reactivity (CVR), delayed-neutron fraction, prompt neutron lifetime and boron influence have to be examined. These in turn are used to gain confidence in the possible viability of this kind of reactor and to meet current licensing standards, before detailed safety studies based on neutron kinetics.

III.B.1 *Fuel temperature coefficient of reactivity (Doppler coefficient)*

The Doppler coefficient defined as :

$$\alpha_{fuel} = \frac{d\rho}{dT_{fuel}} \mid_{T_{mod}} \text{(pcm/K) with } \rho = \frac{k_{inf}-1}{k_{inf}}$$

represents the change in reactivity due to fuel temperature variation. It is calculated for an infinite assembly. This fuel temperature coefficient has been estimated at 1000K for a PWR by using k_{inf} values at 900K and 1100K and at 900K for a CANDU by using 800K and 1000K k_{inf} values.

In PWR, Doppler coefficients are always negative at BOC and EOC : between -2.9 and -3.3 pcm/K for Th/Pu classical fuel (10%(at) and MR=2), between -1.7 and -2.9 pcm/K for UOX fuel, and between -1.9 and -2.9 pcm/K for Th/U-233 fuel (3.2% (at) and MR=2). Under-moderated thorium-based core Doppler coefficients are found to be more negative. A possible explanation is that this kind of faster spectra makes the Th-232 resonances more effective regarding the global Doppler coefficient.

The CANDU values are reported in Table 4.

TABLE 4 – Impact of the MR and of the fissile proportion in fuel on the CANDU Doppler coefficients (with DRAGON)

Fuel Type (at %) / MR	α_{fuel}^{BOC}	α_{fuel}^{EOC}
Unat 0.7% / 15.6	-1.18	0.38
Th/Pu 1.8% / 15.6	-0.58	-0.23
Th/Pu 2.2% / 30	-0.24	0.21
Th/Pu 2.2% / 15.6	-0.68	-0.37
Th/Pu 2.2% / 10	-1.33	-1.17
Th/Pu 3.0% / 15.6	-0.82	-0.75
Th/U3 1.5% / 30	-0.66	-0.71
Th/U3 1.5% / 15.6	-0.96	-1.05
Th/U3 1.5% / 10	-1.42	-1.56

In the more thermal spectrum of CANDU, the resonance broadening due to temperature increase has a smaller impact on reactivity than in PWR. Evolution of Doppler coefficient with burnup is attenuated in Th/U-233 and Th/Pu core compared to the big variation in Unat core. In Th/Pu core, U-233 buildup and Pu-239 burnup could tend to soften change in Doppler coefficient given that U-233 resonance absorption is lower than Pu-239 one, but that remains to be verified.

Contrary to PWR, Doppler coefficients increase with burnup. The reason is that in CANDU, the positive contribution of fissions to reactivity increases faster with burnup than the negative effect of captures. MR variations show similar effects as for PWR. Let us note that a positive Doppler coefficient does not necessarily mean a less safely operating core. Doppler calculations have to be done on an entire core to average new and old bundles in terms of reactivity.

III.B.2 *Coolant void reactivity*

Feedback void coefficients could be calculated by continuously decreasing the moderator density. We rather deal here with the overall balance :

$$\alpha_{void} = \Delta\rho = \frac{k_{inf}(d_{cal}) - k_{inf}^{ref}}{k_{inf}(d_{cal}) * k_{inf}^{ref}} \text{ (pcm) with}$$
$$d_{cal} = 0.0g/cm^3$$

Such a value, calculated for a complete core, stands for a total loss of coolant (LOCA). This coefficient does not symbolize a gradual decrease in reactor coolant flow rate but is relevant for an accident which postulates a break in the feedwater line. Precise kinetic transient analysis would be necessary to evaluate precise transient behaviour anyway.

In PWR, every cases studied have a negative CVR, with a closest value to zero of -16000 pcm. As for CANDU, CVRs are included between 900 and 1400 pcm. These values remain typical of such reactors.

III.B.3 *Delayed-neutron fraction*

The effective delayed-neutron fraction β_{eff} is determined at BOC for each case from individual nuclei reference values Ref. [11] and macroscopic fission cross-sections. The delayed-neutron fraction of a specific fuel almost remains the same throughout core modifications. Typically, for both reactors, the delayed-neutron fraction is around 300 pcm for Th/Pu and Th/U-233 fuels, 550 pcm for U/Pu fuels and 740 pcm for UOX and Unat fuels (which fits with reference values of Ref. [12]). No really significant shift can be noticed but it is seen in PWR that the delayed-neutron fraction is slightly reduced with the MR increase. Indeed for high MR, spectrum is more thermal and we have checked that Th-232 contribution to β_{eff} decreases enough to significantly drop the delayed-neutron fraction : in a Th/Pu (10% at) PWR assembly, for a MR equal to 1.29, β_{Th232} is equal to 16% of β_{eff} (β_{Pu239}=46% of β_{eff}) and for a MR of 3.08, β_{Th232} is equal to 9% of β_{eff} (β_{Pu239}=53% of β_{eff}).

III.B.4 *Prompt neutron lifetime*

The prompt neutron lifetime (l_p) at BOC as determined by MCNP is shown in Tables 5 and 6 for several fuel types. The values are calculated for an infinite assembly.

TABLE 5 – PWR prompt neutron lifetimes at BOC

Fuel Type (at %) / MR	l_p (μs)
Th/Pu 10% / 3.08	8.04
Th/Pu 10% / 2.01	5.23
Th/Pu 10% / 1.29	3.52
Th/U3 3.2% / 3.08	28.7
Th/U3 3.2% / 2.01	19.8
Th/U3 3.2% / 1.29	13.4
UOX 3.5% / 2.01	19.9

TABLE 6 – CANDU prompt neutron lifetimes at BOC

Fuel Type (at %) / MR	l_p (ms)
Th/Pu 2.2% / 30	1.21
Th/Pu 2.2% / 15.6	0.520
Th/Pu 2.2% / 10	0.282
Th/U3 1.5% / 30	1.42
Th/U3 1.5% / 15.6	0.639
Th/U3 1.5% / 10	0.363
Unat 0.7% / 15.6	0.874

Observations are very similar for both types of reactor. In general, under-moderated and highly enriched reactors would be harder to control. Besides, significant discrepancies occur between fuel types. For instance, the Th/Pu PWR design has a prompt neutron lifetime more than three times lower than those of Th/U-233 and UOX assemblies, which tends to increase the nervousness of the core.

III.B.5 *Soluble boron influence*

In order to verify our hypothesis postulating that boron impact on the fuel isotopic vector is negligible, discrepancies on inventories (between calculations made with and without boron) have been computed at EOC. This analysis points out that relative differences on main isotopes (Th-232, Pu, U-233, U-235 and U-238) are not higher than 4%. On other isotopes, differences could be up to 26% but it concerns isotopes like Np-239 whose amount is very low. In order to reduce these inventory differences, calculations with an average boron concentration have been performed, halfening this way the discrepancies.

The boron efficiency depends on the fuel type. For example a Th/Pu core would need more dissolved boron than a Th/U3 one for which the boron efficiency is equal to -11.49 pcm/ppm (ppm of natural boron mass) (BOC) and -8.67 pcm/ppm (EOC). Moreover a low MR affects the soluble boron worth. For the Th/Pu under-moderated core case (10% (at) enriched), the boron efficiency is equal to -1.94 pcm/ppm (BOC) and -1.65 pcm/ppm (EOC) whereas for the "normally" moderated equivalent case (MR=2.01), the boron efficiency is equal to -3.48 pcm/ppm (BOC) and -3.03 pcm/ppm (EOC). Low moderation reduces boron efficiency more than by half.

Concerning the feedback coefficients, soluble boron sometimes has a significant influence. The Doppler temperature coefficients, the delayed-neutron fraction and the prompt neutron lifetime show no real significant changes. However high levels of dissolved boron cause a more-positive moderator temperature coefficient Ref. [13].

IV SCENARIO STUDIES

In the previous section, we have studied the use of Thorium in conventional and slightly modified water reactors. In scenario studies, the most interesting features of these modified water reactors can be exploited, depending on a prioritary goal (fissile production for GenIV startup, waste reduction ...). Many technology development strategies (no recycling, monorecycling, introduction date of new reactors, fleet total power capacity over time, reprocessing time, new reprocessing technology availability dates ...) can also be taken into account. Before describing the scenarios we have studied and their results, we introduce the new scenario code OSCAR that we have developed.

IV.A OSCAR

A new code named OSCAR (Optimized Scenario Code for Advanced Reactors) has been developed to simulate fleet scenarios. It is designed to simulate equilibrium and transition scenarios. The main idea is to let the code optimize each scenario according to user-defined constraints.

The main algorithm can be summarized as follows :

- For the initialization, fuel and reactor data are stored, and the simulation options are read together with the initial fleet and stocks.

1. At each time step, the fuel production of last step is first added to the *available* stock (note that a cooling time may include a delay).
2. Then reactors are aged up, and those reaching their lifetime are removed from the fleet. The reactors imposed by the user are added to the fleet.
3. The necessary fuel is removed from the available stock to run reactors and facilities (fuel fabrication and reprocessing). If fuel is missing for one type of reactor, the simulation goes back in time when the newest reactor was built and restarts with a limited power for this reactor at this step (return to 1 for a previous time step).
4. If there is enough fuel for all reactors (and facilities) at that time, the last part of the algorithm consists of building the missing power of the fleet. In such a case, the choice of the new reactors is automatically done according to the reactor technology availability, to the reactor type priority and to the fuel availability (note that input fuels are directly removed when a reactor is chosen).
5. Once each time step is validated, and until the full scenario end, the current fleet and available stockpile are stored for results analysis and potential back-up in time.

Let us now detail several important issues here, for a better understanding of the OSCAR way of scenario study. First, the reactor priorities are automatically adjusted when necessary. A priority is decreased if a back-up in time is imposed by a specific reactor, and is reset to its original value if the input fuel is made available.
Secondly, some fuel may be produced by a facility (such as an enrichment plan), but the quantity needed is usually not known in advance (during a transition phase). In order to manage such variable needs in fuel facilities, we introduced in OSCAR the possibility of post-built facilities with automatic capacity.

Finally, even if the code is designed for an *automatic optimization* of scenarios, many parameters can be preset by the user in order to take into account political, economical, strategic or technical anticipated changes. For exemple, the quantities of fuel getting in & out or the priority can be changed at a specific date for a type of reactor. A reactor construction can be imposed at a specific date.

From a more practical point of view, this new code named OSCAR is a C++ open-source and universal scenario code. Simulations are handled using a Graphic User Interface, developed using the ROOT package Ref. [14]. This GUI can be used to "modify" and view the input files, and is also very useful to analyse the results. The previous options above described show that the original requirements to handle complex transition scenarios have been fullfilled. The code is also thought to make it as much user-friendly as possible, and some shortcuts are available for simple scenarios. A generic logic has been developed to introduce future options very easily.

IV.B Hypotheses

Transition scenarios are now presented for the French fleet case. A constant 60GWe fleet is supposed. The comparison among the studied scenarios is carried out in terms of natural resource consumption during and after the transition. The main idea of the studied transitions is to produce Plutonium in the existing PWR, and then to use it as a fissile fuel with Thorium to gather U-233, which is the fuel for the GenIV Molten Salt Reactor (MSR). If other "classical" PWRs are required for the transition to take place, UOX fueled EPRs only are used. The following assumptions are made for the scenario studies :

- The U-235 concentration of depleted uranium is set to 0.25% after 2015 for the enrichment process.
- The precise definitive GenIV reactor designs of the MSR and the FBR are not fixed yet. Thus we have reduced their design to a simple isogenerator reactor black box. They are then characterized by their initial fissile inventory : 12t/GWe of Pu for the FBR and 3 or 5 t/GWe of U-233 for the MSR.
- Both types of GenIV reactors become technologically available from 2035.
- The spent fuel of the transition reactors are reprocessed to recover only Uranium + Protactinium and Thorium. The Pu content is considered as waste, thus this fuel is considered as a once-through cycle from the Pu point of view.
- "Classical" PWRs are using only UOX fuel after 2015.
- Transition, EPR and GenIV reactors are assumed to have a 60-year lifetime.

The following scenarios are considered :

1. PWR-only : this is the only technologically feasible solution which is available now. Moreover the recycling is stopped after 2015 for proliferation considerations (there is no use of Pu for another reactor). (PWR-only case)
2. PWR+FBR : PWRs are replaced by EPRs between 2020 and 2035. Then the FBR becomes the only reactor built after 2035. (FBR case)
3. PWR+PWR(Th/Pu)+MSR(3t) : All the Pu content is used to produce U-233 in the PWR transition reactors. MSR are built in priority after 2035. We assume that the initial U-233 inventory to start them is 3t. Similar scenarios were studied by Ref. [3] in which the proportions of Th/Pu fuel were manually adjusted to match the final U-233 needs of the MSR. (PWR_t-MSR case)
4. PWR+PWR(Th/Pu)+PWR(Th/U-233) : All the Pu content is used to produce U-233 in the transition reactors. If no GenIV reactors are built, this case illustrates the transition to a symbiotic fleet of PWR reactors with a reduced uranium requirement.(Symbiotic PWR case)
5. PWR+CANDU(Th/Pu)+MSR(3t) : This case is similar to the case 3 using a CANDU as a Th/Pu transition reactor instead of PWR. (CANDU_t-MSR-1 case)
6. PWR+CANDU(Th/Pu)+MSR(5t) : This case is similar to the previous except for the initial fissile loading of the MSR. It can be seen in Ref. [15] that the design of this kind of reactor may vary according to the different technology development. Thus, we have studied a transition for a larger U-233 stock requirement. This case has not been considered for PWRs because of their lower U-233 productivity. (CANDU_t-MSR-2 case)
7. PWR+CANDU(Th/Pu)+CANDU(Th/U-233) : If no GenIV reactor are built, this case illustrates the transition to a symbiotic fleet of CANDU reactors with a reduced uranium consumption.(Symbiotic CANDU case)

The Th/Pu transtion reactors are the (Th/Pu(10% at) MR3.08)PWR and (Th/Pu(3% at) MR15.6)CANDU reactors. They have been chosen for their good U-233 outputs (see Tables 2 and 3). To have the possibility of heterogeneous core loading, even if this was not simulated in our study, a constraint is added. In the PWR case, the vessel of the core is similar independently of the modifications. Thus any (Th/U-233)PWR transition reactor could have been chosen. However, in the CANDU case, the vessel geometry depends on the MR. Thus we have chosed a (Th/U-233)CANDU transition reactor with the same MR than for the (Th/Pu)CANDU transition reactor. The Th/U-233 transition reactors are the (Th/U3(3.2% at) MR1.29)PWR and (Th/U3(1.5% at) MR15.6)CANDU for their good combined CR and burnup performance.

In the case of Pu multirecycling in the Th/Pu transition reactors, the choice of the reactors may have been more driven by the conversion ratio (CR) instead of the conversion factor (CF).

IV.C Results and Discussion

Results of the different scenarios described earlier are presented on Fig. 6. Calculations have been performed over 160 years for all transitions to completely take place. Even if there are many uncertainties for such a long period concerning the uranium availability (resources and price), Fig. 6 clearly illustrates the trends of Tab. 2 and 3.

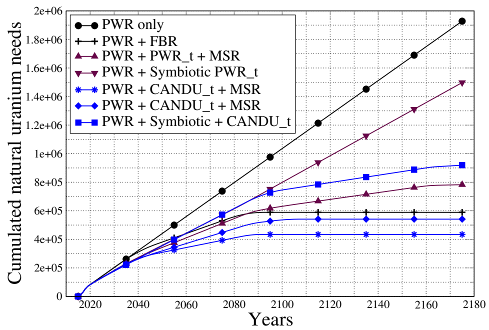

FIG. 6 – Cumulated natural Uranium resource needs for differents transition scenarios

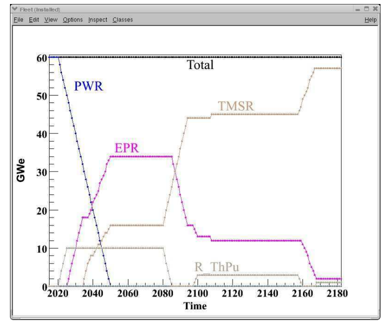

FIG. 7 – Transition scenario for the PWR_t-MSR case, direct output of the OSCAR GUI.

The limit in natural uranium consumption reached by the scenarios using GenIV reactors is very different from one case to another. The FBR case has the fastest transition, but the amount of natural uranium used is higher than for CANDU_t-MSR cases. In fact it can be explained by the good CF conversion factor of Th/Pu transition CANDU reactors combined with the smaller amount of fissile materials required for the MSR than for the FBR. Moreover, the FBR case requires depleted uranium resources which are not unlimited either on a long term, and this may imply additionnal natural Uranium needs. In the PWR_t-MSR case, the transition is very long (See. Fig. 7). In fact, the CF is too small even combined with the 3t of U-233 MSR to be competitive with the FBR case.

Symbiotic and PWR-only cases have no upper limits for natural uranium consumption but they reach an equilibrium as expected. The PWR-only and PWR-Symbiotic cases have a very large amount of cumulated natural Uranium resource needs and a large natural Uranium consumption per year. Compared with this two cases, the CANDU-case is very sparing in term of natural Uranium needs, but it is still more natural Uranium consumming than any GenIV case. The performance of the symbiotic and PWR-only cases is directly related with the installed power from EPRs at equilibrium. In the 60GWe French fleet case, it is 60GWe for the PWR-only case, 48GWe for the PWR-Symbiotic case and only 7GWe for the CANDU-Symbiotic case. The EPR installed power is sensitive to the CF of the Th/Pu transition reactor and very sensitive to the U-233 economy of the Th/U3 transition reactor. In the CANDU-Symbiotic case, it could even lead to a 100% CANDU-Th/U-233 fleet if a self-sufficient cycle is obtained (see Sect. III.A).

Finally, no case takes into account the evolution of the fuel due to multi-recycling of the Pu for the FBR case or of the Uranium for the MSR cases. This could also have an impact on the symbiotic fleet or on the transition time and should be investigated in the next studies.

V CONCLUSIONS

In this paper, several options have been investigated for exploiting the ability of PWR and CANDU reactors to burn a variety of fuels. The methodology (currently core-equivalent assembly) has been improved from simple cell calculations and bundle level methods Ref. [16]. However some points need more improvements : for example, our PWR kinf linearity hypothesis meets its limit with Th/U-233 fuels.Moreover, in order to determine if Th/Pu and Th/U-233 cores would meet safety operating criteria, the behaviour of our fuel/reactor systems needs to be more thoroughly checked by detailed neutron kinetics studies. This exploratory work gives anyway some basic trends on the high conversion potential of such systems. Simple reactor modifications enable us to determine which feasibility area is conceivable and which performance thorium-based fuels can achieve in slightly modified PWR and CANDU reactors.

Our core-equivalent assembly calculations here have obviously prevented us from reaching the highest CRs. Further core optimizations are necessary, especially for the PWR. It appears more clearly to us now that optimal conversion goes through core heterogeneity on one hand, and through spectrum shift techniques on the other hand. These latter have been for instance conceived for the Framatome RCVS, and even used on the Shippingport Light Water Breeder Reactor Ref. [17]. The main practical idea that made this LWBR breed U-233 over a few year period leans on a movable core geometry, used as a neutron economical way of long-term reactivity management. MURE flexibility will hopefully help us to simulate such complex techniques. Finally, detailed core calculations will allow us to evaluate the possible gains on CR obtained from optimized fertile/fissile spatial distributions, as proposed for example by the Radkowski seed-blanket core design Ref. [10].

The results of the scenario studies show that many transitions are possible. Independently of the individual performance of each reactor, their integration into the fleet may not be obvious. The automatic optimization performed by OSCAR is very useful from this point of view, and can reveal promising scenarios like symbiotic ones.

The scenario studies were compared only in terms of natural Uranium resource needs. It could be interesting to present also the waste production, the reprocessing needs, the fissile material inventory and to estimate the cost of the transition. It can easily be done with the actual version of the code OSCAR and will be presented in more details in another paper together with the optimization process. Finally, the scenario code is still under development, we plan to improve it to take the fuel evolution into account, and to automatically adjust the fuel quantity with the isotopic vector. This may affect the conclusions of the different scenarios regarding the best scenarios.

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